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SUBJECT: Statement of matl facts as to which there is no genuine issue to be heard re Contention 8.

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July 16, 1981

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

JUL 22 1981
Office of the Secretary
Docketing & Service
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In the Matter of	)	
	)	
PENNSYLVANIA POWER & LIGHT COMPANY	)	
	)	
and	)	Docket Nos. 50-387
	)	50-388
ALLEGHENY ELECTRIC COOPERATIVE, INC.	)	
	)	
(Susquehanna Steam Electric Station, Units 1 and 2)	)	

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APPLICANTS' STATEMENT OF MATERIAL
FACTS AS TO WHICH THERE IS NO
GENUINE ISSUE TO BE HEARD (CONTENTION 8)

Pursuant to 10 C.F.R. § 2.749(a) Applicants state, in support of their Motion for Summary Disposition of Contention 8 in this proceeding, that there is no genuine issue to be heard with respect to the following material facts:

1. Section 5.3.3, Part II.6 of the Commission Staff's Standard Review Plan ("SRP") sets forth a recommended criterion for acceptable reactor vessel behavior in the event of thermal shock to the vessel caused by emergency core cooling system operation after a loss-of-coolant accident. The criterion is that the vessel must remain leaktight enough to support adequate core cooling after such thermal shock. Analyses based on generally accepted principles and procedures of linear elastic fracture mechanics are adequate to demonstrate conformance with the criterion. SRP, § 5.3.3, Part II.6.

2. General Electric Company ("GE") has performed analyses to verify that the reactor pressure vessels for the Susquehanna Steam Electric Station ("Susquehanna") meet the thermal shock criterion

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specified in the SRP. These analyses were performed in accordance with the American Society of Mechanical Engineers' Boiler and Pressure Vessel Code ("the Code"). Affidavit of John F. Copeland and Franklin E. Cook in Support of Summary Disposition of Contention 8 ("Copeland-Cooke Aff."), Para. 3.

3. To assure that a reactor vessel will not crack under the thermal shock of cool ECCS water after blowdown, it is necessary to determine that the particular combination of stress intensity and temperature needed to cause crack propagation will not occur. Copeland-Cooke Aff., para. 4.

4. When cool ECCS water reaches the reactor vessel, stresses are created in the vessel wall because the inside surfaces cool faster than the outside surfaces. The intensity of these stresses is calculated using a finite element program and standard mechanical engineering procedures and the highest stress location is determined. As required by the Code, a one-quarter through wall flaw is then postulated at this location; the flaw serves to concentrate the stresses at that location. Using multipliers specified in the Code, these stresses are translated into a stress intensity factor representing the concentration of the stress at the location of the flaw. The minimum reference nil ductility transition temperature ("minimum RT_{NDT}") is then determined from a curve, derived from many years of test values and presented in the Code. Id., para. 5.

5. Finally, the reference nil ductility transition temperature for the reactor vessel material ("material RT_{NDT}") at the

critical location is ascertained as specified in the Code. If the material RT_{NDT} is lower than the minimum RT_{NDT} the postulated flaw will not propagate from the stress of cool ECCS water being injected into the vessel. Copeland-Cooke Aff., para. 5.

6. For the Susquehanna reactor vessels, the temperature of the ECCS water when it reaches the vessel wall was conservatively assumed to be 50°F. This is the lowest temperature conceivable for the ECCS water, which will come from sources inside the facility for the duration of the thermal shock condition, will pass through pipes and pumps before reaching the vessel, and will impact the core shroud before impacting the vessel wall. Id., para. 6.

7. The stresses on the Susquehanna vessels caused by injection of 50°F ECCS water were determined based upon detailed stress analysis for a reactor vessel fabricated of the same materials as those for Susquehanna, with essentially identical wall thicknesses and similar basic dimensions. This analysis showed that for the Susquehanna vessels, the reactor vessel bottom head was the critical location and that an RT_{NDT} for the bottom head material of 94°F or lower will assure that crack propagation does not occur on ECCS injection. Id., para. 7.

8. The material RT_{NDT} for a vessel is obtained by testing the material prior to vessel fabrication. Because the RT_{NDT} concept was not added to the Code until after the Susquehanna vessels were designed, the materials obtained and tested, and fabrication started, test results for the Susquehanna vessel material had to be correlated to the RT_{NDT} definitions. As correlated, conservative

values for RT_{NDT} of the Susquehanna bottom head material are 34°F for Unit 1 and 20°F for Unit 2, well below the 94°F RT_{NDT} at which crack propagation would be predicted to occur. Id., para. 8. Therefore, the analysis shows that the Susquehanna reactor vessels will survive, with considerable margin, the thermal shock of cool ECCS water after blowdown without cracking. Id., para. 3.

10. Although RT_{NDT} for reactor vessel material increases with neutron irradiation of the material and therefore increases over the life of the plant, the bottom head material is not subject to sufficiently high neutron fluences for the RT_{NDT} to shift. Other areas of the vessel, such as the beltline, which experience higher neutron fluences are not subjected to thermal shock since ECCS coolant water is not injected into the core beltline region in BWR's such as Susquehanna. Id., para. 9. Therefore, the considerable margin existing in the Susquehanna vessels against thermal shock will be preserved for the life of the units.

Dated: July 16, 1981.

Respectfully submitted,

SHAW, PITTMAN, POTTS & TROWBRIDGE

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